ACCESSION #: 9907060154

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Joseph M. Farley Nuclear Plant - Unit 1 PAGE: 1 OF 4

DOCKET NUMBER: 05000348

TITLE: Unit 1 Reactor Trip Following Loss of the 1A Steam

Generator Feedwater Pump

EVENT DATE: 05/27/1999 LER #: 1999-002-00 REPORT DATE: 06/25/1999

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 92

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: L. M. Stinson, TELEPHONE: (334) 899-5156

General Manager Nuclear Plant

COMPONENT FAILURE DESCRIPTION:

CAUSE: X SYSTEM: JI COMPONENT: MANUFACTURER:

REPORTABLE EPIX: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On May 27, 1999 at 08:55, with turbine load being reduced rapidly as a result of loss of the 1A Steam Generator Feed Pump (SGFP), FNP Unit 1 Reactor tripped on Overtemperature Delta T (OT Delta T). At 08:54:15, while attempting to swap the on service SGFP oil cooler, oil pressure to the SGFP was momentarily lost resulting in a trip of the SGFP. In response to the SGFP trip, the operator performed a rapid manual turbine

load reduction. Control rods inserted in automatic; however, due to the magnitude of the power mismatch, Reactor Coolant System (RCS) temperature rose rapidly and resulted in the OT Delta T Reactor Trip setpoint being reached. A pressurizer Power Operated Relief Valve (PORV) lifted prior to the trip and then closed as designed.

This event was due to personnel error by the operator performing an excessive load reduction. This was complicated by the failure of the steam dump system to respond as quickly as designed. The loss of main feedwater abnormal operating procedure has been enhanced to provide additional guidance for a loss of a SGFP event. Operations and licensed personnel have been made aware of these changes. These procedure changes will be covered during appropriate simulator training on response to scenarios involving loss of a SGFP.

TEXT PAGE 2 OF 4

Westinghouse -- Pressurized Water Reactor

Energy Industry Identification Codes are identified in the text as [XX].

Description of Event

For several days prior to the event, the 1A Steam Generator Feed Pump (SGFP) [SJ] had experienced a decline in oil pressure. The inline oil filters [SL] had been cleaned but pressure had declined again shortly after cleaning. It was believed that the on-service oil cooler [SL] might be fouled, resulting in higher oil temperature and lower oil viscosity and pressure. Therefore it was decided to swap the on-service oil cooler. The SGFP is designed to allow swapping oil coolers without interruption of oil flow. This evolution had recently been accomplished on the Unit 2B SGFP, correcting an oil temperature problem without incident.

Several attempts to swap the on-service oil cooler were unsuccessful due to difficulty moving the selector valve. The valve stem was cleaned and packing loosened, and on May 27, 1999 at 08:54, mechanical advantage was applied to the valve. The valve moved a small amount and then suddenly gave way moving past alignment to the previously off-service cooler, momentarily interrupting oil pressure and resulting in the 1A SGFP trip.

In response to the SGFP trip, the control room operators placed the turbine control in manual governor valve fast action and began closing the governor valves to reduce turbine load to less than 515 Mwe as directed by the procedure. Boration was initiated, and all Auxiliary Feedwater (AFW) [BA] pumps were started. At 08:54:55 with 545 Mwe indicated generator load, the operator released the governor valve fast action close button. Although 545 Mwe (an approximate 40% load reduction) was indicated, the turbine control system [JJ] [TE] had positioned the valves for an approximate 60% load reduction. This was due to indicated electrical output lagging behind valve demand position. Control rods inserted in automatic as designed.

The Steam Dumps [JI] opened during the load reduction, but did not respond as quickly as designed.

Due to mismatch of reactor power and steam load, the Reactor Coolant System (RCS) [AB] average temperature (Tavg) increased rapidly. At 08:55:08, when the rate of increase had reached approximately 0.5 degrees per second during the load reduction, the reactor tripped automatically on Overtemperature Delta T (OT Delta T). The dominant factor in reducing the OTAT trip setpoint to the actual RCS loop Delta T has been determined to be the penalty for the rate of temperature increase calculated by the lead-lag circuit. Following the trip all rods inserted and safeguards equipment functioned as designed. A pressurizer power operated relief valve (PORV) lifted prior to the trip, and then closed as designed. The steam dump system modulated to control the RCS temperature to no-load Tavg.

Cause of Event

The reactor trip was caused by the operator performing an excessive load reduction, complicated by the failure of the steam dump system to respond as quickly as designed. An engineering review of the behavior of the OT Delta T circuit, PORV, and steam dumps was performed.

The excessive load reduction was caused by personnel error by the operator, complicated by an unclear procedure and inconsistent training on rapid load reduction.

Safety Assessment

All safeguards systems functioned as designed.

The health and safety of the public were unaffected by this event and no release of radioactivity occurred.

The steam dump system is a plant control system designed to lessen the impact of Condition 1 and Condition 2 events. Condition 1 is normal plant operation and Condition 2 describes incidents of moderate frequency such as reactor trips, turbine trips, and other non-LOCA type events. The steam dump system provides a degree of diversity to pressurizer safety valves and the steam generator safety valves. The steam dump system is designed to reduce challenges to these safety valves, thereby reducing the chance that a Condition 1 or 2 event will propagate into a more serious condition as a result of a stuck-open safety valve. No credit is taken for steam dump operation for more serious Condition 3 or 4 events, which bound Condition 1 and 2 events. In addition, the steam dumps are non safety-related. FSAR section 15.2.7.1 describes the effects of failure of steam dump valves to open during loss of load events. In summary, it states the reactor will trip by diverse means and that the pressurizer safety valves and steam generator safety valves are sized to protect the RCS and steam generator against overpressure for all load losses without assuming operation of the steam dump system.

The degraded steam dump capacity represents a reduction in "defense in depth" to RCS and secondary plant overpressurization events due to loss of turbine load. However, sufficient overpressure protection is provided by the safety-related RCS and steam generator safety valves, and therefore the health and safety of the general public is not affected by this condition.

Core damage frequency is not affected by the degraded steam dump capability because sufficient capacity is provided with the remaining dumps to provide for plant cooldown to RHR conditions. In addition, steam generator atmospheric relief valves provide a redundant means for plant cooldown.

Assuming no steam dump capability with no other degraded conditions results in a Risk Achievement Worth (RAW) value of 1.0002, or virtually no impact on core damage frequency.

Corrective Action

The unit was stabilized in Hot Standby. The on-service cooler was swapped and oil strainers were replaced to restore SGFP oil pressure to normal.

The unit was returned to power operation.

The loss of main feedwater abnormal operating procedure has been enhanced to provide additional guidance for a loss of a SGFP event. Operations and licensed personnel have been made aware of these changes. These procedure changes will be covered during appropriate simulator training on response to scenarios involving loss of a SGFP.

Additional Information

The trip of the SGFP was caused by personnel error in that mechanical advantage was inappropriately used on the oil cooler swap valve, resulting in a momentary loss of oil pressure. Management expectations regarding pre-job briefings, job planning, and supervisory oversight have been reinforced. The procedure for swapping oil coolers has been revised to include enhanced guidance on swapping coolers and additional vendor guidance on how to correct valve operating difficulties. The oil cooler swap valve will be repaired during the next outage of sufficient duration. Additional administrative controls have been placed on the operation of this valve until its material condition is corrected.

Failure of the steam dump system to respond as quickly as designed was discovered during a post-trip review of plant parameters. At this time the steam dump system is degraded but operable. Several of the eight steam dump valves have been determined to have slow response times. Further trouble-shooting and repair of these valves, which can be accomplished at power, is in progress. If this trouble-shooting determines valves must be disassembled, disassembly and repair will be performed at the next outage of sufficient duration. An appropriate periodic test procedure will be developed to ensure the valve function is periodically tested.

The above actions will be completed following the next outage of sufficient duration on Unit 1. Testing of the other unit's steam dump valves is being scheduled. A four-hour non-emergency notification was made per 10 CFR 50.72 (b)(2)(ii). The root cause investigation into this event identified a condition of inaccurate simulator modeling of the speed of turbine governor valve response. This has been evaluated and determined not to have been a factor in the operator's inappropriate response to the SGFP trip. Simulator modeling of the governor valve response has been corrected.

In the past two years there have been no reactor trip events due to either personnel error or procedure inadequacy. Therefore no related LERs have been submitted in this period.

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Dave Morey Southern Nuclear

Vice President Operating Company, Inc.

Farley Project Post Office Box 1295

Birmingham, Alabama 35201

Tel 205.992.51 31

SOUTHERN

June 25, 1999 COMPANY

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U. S. Nuclear Regulatory Commission

ATTN: Document Control Desk

Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant - Unit 1

Licensee Event Report 99-002-00

Unit 1 Reactor Trip Following Loss of the

1A Steam Generator Feedwater Pump

Ladies and Gentlemen:

Joseph M. Farley Nuclear Plant - Unit 1 Licensee Event Report No.

99-002-00 is being submitted in accordance with 10 CFR 50.73(a)(2)(iv).

There are no NRC commitments in the Licensee Event Report.

If you have any questions, please advise.

Respectfully submitted,

Dave Morey

EWC/maf:99-02.doc

Enclosure

U. S. Nuclear Regulatory Commission

cc: Southern -Nuclear Operating Company

Mr. L. M. Stinson, General Manager - Farley

U. S. Nuclear Regulatory Commission, Washington, D. C.

Mr. M. L. Padovan, Licensing Project Manager - Farley

U. S. Nuclear Regulatory Commission, Region II

Mr. L. A. Reyes, Regional Administrator

Mr. T. P. Johnson, Senior Resident Inspector - Farley

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